NEUTRONIC SAFETY PARAMETERS AND TRANSIENT ANALYSES FOR POTENTIAL LEU CONVERSION OF THE BUDAPEST RESEARCH REACTOR*

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ABSTRACT

An initial safety study for potential LEU conversion of the Budapest Research Reactor was completed. The study compares safety parameters and example transients for reactor cores with HEU and LEU fuels. Reactivity coefficients, kinetic parameters and control rod worths were calculated for cores with HEU (36%) UAl alloy fuel and UO_2-Al dispersion fuel, and with LEU (19.75%) UO_2-Al dispersion fuel that has a uranium density of about 2.5 g/cm^3. A preliminary fuel conversion plan was developed for transition cores that would convert the BRR from HEU to LEU fuel in 3-4 years after the process is begun.

INTRODUCTION

The neutronic feasibility for potential LEU conversion of the Budapest Research Reactor at the KFKI Atomic Energy Research Institute was studied in a previous report[1]. That report compared the reactor performance with the current HEU (36%) fuels and a LEU (19.75%) fuel. Comparisons were made of the equilibrium fuel cycle lengths, the thermal neutron fluxes in the in-core and ex-core experiment locations, and the control- and safety-rod reactivity worths. Those results showed that conversion of the BRR from HEU fuel to LEU fuel is feasible if a qualified LEU fuel is available and if all safety criteria are satisfied.

The calculations in this paper have been performed to determine the neutronic safety parameters of the BRR with these same HEU and LEU fuels and to compare the response of the reactor to several example transients. The two current HEU fuel assembly types include a 36.8% enriched UAl alloy-fuel assembly (VVR-SM) with 40.2 g ^{235}U and a 36.2% enriched UO_2-Al dispersion-fuel assembly (VVR-M2) with 44.3 g ^{235}U. The main LEU fuel assembly that was studied here has 52.3 g ^{235}U in 19.75% enriched UO_2-Al dispersion fuel. This report also includes results for a LEU fuel assembly design with 50.0 g ^{235}U. Comparisons are made of reactivity coefficients, kinetic parameters, critical control-rod positions, safety-rod reactivity worths, and reactor shutdown margins. The study includes several transient analyses using the calculated safety parameters. A scenario to convert the BRR from HEU fuel to LEU fuel has also been studied.

REACTOR DESCRIPTION

Detailed descriptions of the 10 MW, light-water cooled BRR and its VVR-type fuel assemblies were given in Ref. 1. A representative model of the reactor core with the fuel assemblies, experiment

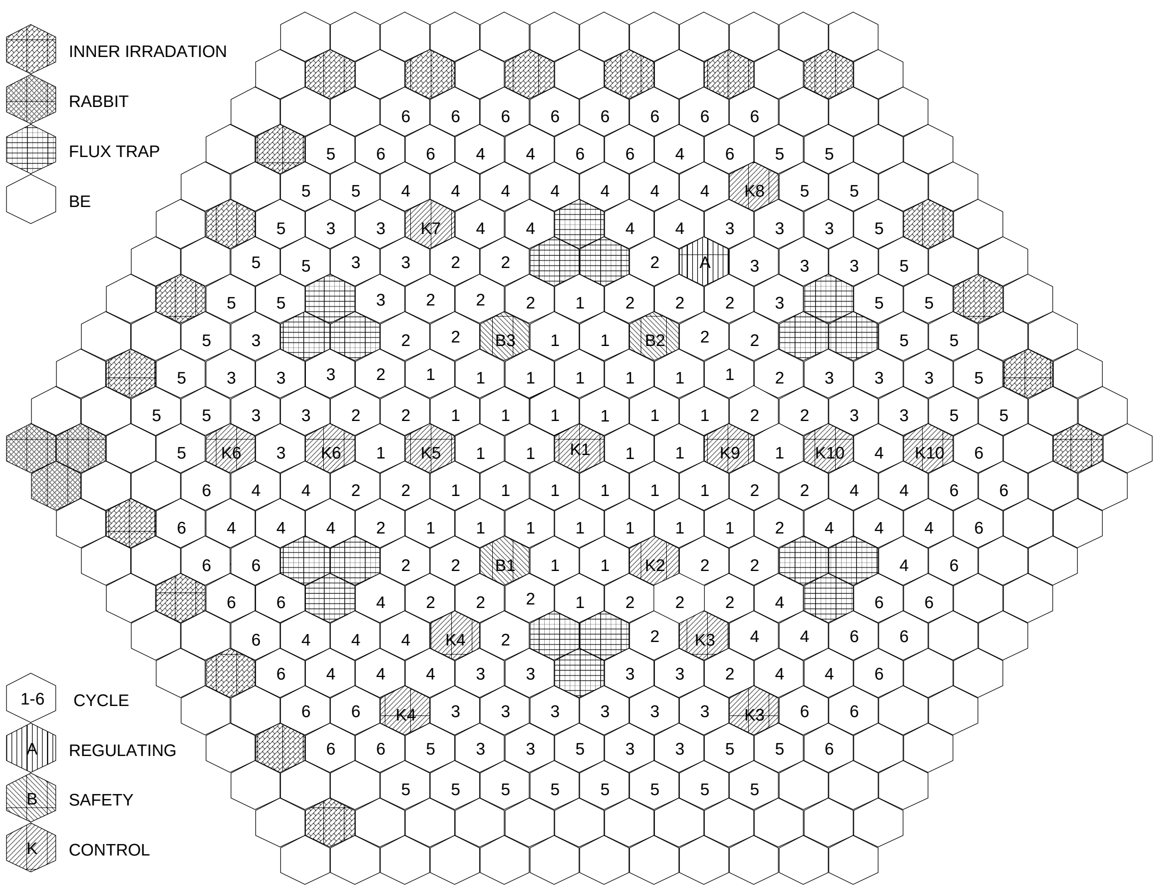

Figure 1. BRR Equilibrium Core Configuration (228 Fuel Assemblies)

locations, and reflector is shown in Fig. 1. The reactor core configuration, based upon recent fuel management,[2] contains about 228 fuel assemblies and is a mixture of the two HEU fuel assembly types at various stages of burnup. The reactor assemblies can be roughly divided into six groups of 38 assemblies, each with about the same level of burnup. The BRR is operated in a semi-equilibrium fuel cycle mode in which burned fuel is moved outwards, spent fuel is discharged from the edge, and fresh fuel is inserted at the center. This mode of operation also permits flexibility to configure the core and the experiments in a preferred way. An idealized equilibrium fuel cycle model of the BRR core is that shown in Fig. 1.

This model shows the fuel assembly locations (3.5-cm pitch) and the six burnup-level groups (1 - 6), the control (K*n*), safety (B*n*) and regulating (A) rod locations, the in-core and ex-core experiment locations, and the replaceable beryllium reflector assemblies that surround the reactor core. There are 14 control rods located in hex-rings 1, 4, 6 and 8, three safety rods in ring-4, and the regulating rod in ring-6. The rods move from 5 cm below the core (inserted) to 5 cm above the core (withdrawn) in a 70-cm stroke over the 60-cm height of the active core.

Surrounding the replaceable Be reflector is a fixed Be reflector, an aluminum reactor tank, a light-water reflector, and another aluminum reactor tank. In addition, there are ten horizontal beam tubes (8 radial and 1 tangential) and a cold neutron source tube located on the reactor midplane. The beam tubes extend to a point near the fixed-Be / replaceable-Be interface boundary.

CALCULATIONAL METHODS

The beginning of an equilibrium fuel cycle (BOEC) was modeled for most reactor calculations performed in this study; for some calculations, however, a hypothetical fresh fuel, 228-assembly reactor core was also modeled. All neutronic safety parameter results were obtained using the DIF3D multigroup diffusion theory code[3], which uses group-dependent internal boundary conditions on absorber surfaces to model the control, safety and regulating rods[4]. The internal boundary conditions for the B_4C absorber materials were calculated using the MCNP Monte Carlo code[5]. When rods were modeled in DIF3D, only the finite-difference solution option was used. When rods were not modeled, either the finite-difference or the nodal[6] solution option in DIF3D was used.

Burnup calculations were performed using the REBUS-3 code[7] and kinetic parameters were calculated using the MC^2-2 code[8] and the VARI3D code[9]. The WIMS-ANL code[10] was used to generate multigroup cross sections in seven broad energy groups based on a WIMS 69-group library prepared at ANL using ENDF/B-VI nuclear data. Transient analyses were performed using the RELAP5 code[11] and the PARET code[12].

FUEL CONVERSION STUDY RESULTS

Equilibrium Fuel Cycle

The equilibrium fuel cycle lengths of the BRR with each of the fuel assembly types were calculated and reported in Ref. 1. The fuel cycle length utilizing a 19.75% enriched, 2.47 gU/cm^3 UO_2-Al dispersion-fuel assembly with 50.0 g^{235}U per assembly has also been calculated here. This assembly loading is 1.2 times the loading (41.7 ± 2.1 g^{235}U) of the VVR-M2 type experimental assemblies currently under irradiation testing[13] in the VVR-M reactor at the Petersburg Nuclear Physics Institute in Russia. The 1.2 factor is the ratio of the BRR fuel assembly active length (60-cm) to the VVR-M fuel assembly active length (50-cm). Table 1 shows the calculated fuel cycle characteristics of the HEU and LEU fuels in the BRR.

Table 1. Equilibrium Fuel Cycle Characteristics for the Reactor Model With 228 Assemblies and Six Burnup-Level Groups of 38 Assemblies Each

Fuel Assembly Type	Enr., % / ^{235}U Mass, g	Equilibrium Fuel Cycle Length, d	Average ^{235}U Discharge Burnup, %	Fuel Assemblies Used per Year[a]
HEU UAl Alloy	36.8 / 40.2	84.5	67.4	75
HEU UO_2-Al	36.2 / 44.3	98.8	70.8	64
LEU UO_2-Al	19.75 / 50.0	120.1	73.2	53
LEU UO_2-Al	19.75 / 52.3	127.8	74.4	50

[a] Assumes 4000 full power hours or 167 full power days of operation per year.

These data show that decreasing the LEU fuel assembly loading from 52.3 to 50.0 g^{235}U reduces the fuel cycle length from 128 days to 120 days and increases the number of LEU assemblies used per year from 50 to 53.

Transition Core Conversion Studies

The purpose of these studies is to show that conversion of the BRR from HEU to LEU fuel can be accomplished in an efficient and safe manner. A preliminary scenario for the conversion has been developed for the reactor model shown in Fig. 1. This model has six burnup-level groups of 38 assemblies each and assumed HEU (UO_2-Al, 44.3 g^{235}U) and LEU (UO_2-Al, 52.3 g^{235}U) fuels. A final conversion plan will need to be developed based upon the reactor configuration at the time of conversion and actual HEU and LEU fuels. In order to fully utilize the current HEU fuel and to maintain the current equilibrium fuel cycle characteristics, the plan is to introduce fresh LEU fuel instead of fresh HEU fuel at the beginning of each fuel cycle. Since there are six burnup-level groups of fuel, six transition cores would be necessary over about three years to fully convert from HEU to LEU fuel.

The first transition core would have fresh LEU fuel in burnup-level group-1, and partially burned HEU fuel in groups- 2 to 6. The second transition core would have fresh LEU fuel in burnup-level group-1, partially burned LEU fuel in group-2, and partially burned HEU fuel in groups- 3 to 6. This initial scenario would convert the BRR from HEU to LEU fuel over six fuel cycles. Table 2 shows the calculated BOC and EOC eigenvalues of each transition core and the accumulated reactivity change from the equilibrium HEU-fuel core through transition core-6. For comparison, the eigenvalues of the HEU and LEU equilibrium fuel-cycle cores are also shown. These calculated eigenvalues assume equilibrium fuel-cycle fission product concentrations. Control, safety and regulating rods are fully withdrawn.

Table 2. Transition Core Eigenvalues
(Accumulated Reactivity Change Relative to HEU Equilibrium Core.)

Core Configuration[a]	BOC k-effective	EOC k-effective
HEU Equilibrium Core, UO_2-Al 44.3 g^{235}U	1.1184	1.0202
Transition Core-1[b]	1.1235 (0.40%)	1.0303 (0.95%)
Transition Core-2	1.1299 (0.91%)	1.0431 (2.15%)
Transition Core-3	1.1368 (1.45%)	1.0536 (3.10%)
Transition Core-4	1.1443 (2.02%)	1.0649 (4.11%)
Transition Core-5	1.1509 (2.53%)	1.0747 (4.97%)
Transition Core-6[c]	1.1580 (3.06%)	1.0666 (4.26%)
LEU Equilibrium Core, UO_2-Al 52.3 g^{235}U	1.1228	1.0199

[a] All core configurations have 228 fuel assemblies, divided into six 38-assembly burnup-level groups. The cycle lengths for all core configurations with HEU fuel are 98.8 days, and 127.8 days for the transition core-6 and the equilibrium LEU core.

These data show that the core reactivity increases in time since the reactivity of the LEU fuel with 52.3 g^{235}U per assembly is greater than that of the HEU fuel. The HEU fuel cycle time (98.8 d) is maintained during the conversion transition because full utilization of the excess reactivity available for burnup would result in exceeding the BRR ^{235}U burnup limit[2] of 70% for the HEU (36%) fuel. The calculated average discharge burnup (see Table 1) of the LEU fuel with 52.3 g^{235}U is about 74%.

Transition core-6 has the largest accumulated excess reactivity change (3.06% Δk/k[2]) at BOC relative to the HEU equilibrium core. This amount of reactivity however, can just be controlled using the control rods. The calculated critical control-rod banked position is 12.9 cm with the regulating and safety rods withdrawn. The total worth of the three safety rods is 7.36% Δk/k[2] and the reactor shutdown margin is 2.55% Δk/k[2]. The required minimum shutdown margin[2] is 2% Δk/k[2] with the control and regulating rods inserted and the safety rods withdrawn. Shutdown margins from Ref. 1 for the HEU and LEU equilibrium cores are respectively, 8.29 and 6.21% Δk/k[2].

The reactivity data for the equilibrium and transition cores of Table 2 have assumed equilibrium fuel-cycle concentrations of fission products in burned fuel assemblies and no fission products in fresh fuel assemblies. While the shutdown margin may be satisfied under normal reactor operation, the shutdown margin may be reduced during reactor startup when short-lived fission products have decayed in the burned fuel. For transition core-6, the reactivity effect of the short-lived

^{135}I, ^{135}Xe and ^{149}Pm fission products are calculated to be about 2% $\Delta k/k^2$. This added increase in excess reactivity would result in an unacceptable shutdown margin.

Modification of this six-step conversion scenario is necessary to meet the BRR shutdown criteria at each step. The modified conversion plan will depend upon the reactor core configuration at the time of conversion, the experiment locations, the mix of 36% enriched VVR-SM and VVR-M2 fuel assemblies and the 19.75% loading of the LEU fuel assembly. The excess reactivity of transition cores with 38 assemblies per burnup-level group will also have to be adjusted. Removing fuel from the core periphery to reduce the core size will reduce the excess reactivity. The 38-assembly, six burnup-level groups can be reduced to 37 or fewer assemblies which will remove multiples of six fuel assemblies on the core periphery. After fuel conversion, the reactor core could be adjusted back to a nominal 228-assembly core configuration with a LEU cycle length of 120 - 128 days instead of the current HEU cycle length of 85 - 99 days.

Reactivity Coefficients

The BRR reactivity coefficients with the two HEU fuel types and the LEU fuel assemblies with 52.3 g^{235}U per assembly are shown in Table 3. These data were calculated for an all fresh-fuel reactor core. For comparison, estimates of the reactivity coefficients at the beginning of an equilibrium fuel cycle were also calculated assuming burnup independent cross-sections.

Table 3. Reactivity Coefficients per Degree Centigrade ($\Delta k/k^2$/°C at 60°C)
(All fresh fuel, 228-assembly core configurations with fresh-fuel cross sections.)

Coefficient	HEU-UAl 40.2 g^{235}U	HEU-UO_2 (BOEC)[b] 44.3 g^{235}U	LEU-UO_2 52.3 g^{235}U
Fuel Temperature	-0.99e-5	-1.06e-5 (-1.26e-5)	-1.62e-5
Coolant Temperature	-6.39e-5	-5.64e-5 (-9.37e-5)	-4.64e-5
Coolant Density	-4.49e-5	-4.73e-5 (-8.09e-5)	-5.04e-5
Sum	-11.9e-5	-11.4e-5 (-18.7e-5)	-11.3e-5
Coolant Void[a]	-8.76e-4	-9.24e-4 (-15.9e-4)	-9.85e-4

[a] Reactivity coefficient per percent change in density ($\Delta k/k^2$/% at 60°C).
[b] Equilibrium core configuration at BOEC with burned-fuel (32.5% ^{235}U burnup) cross sections; core-average ^{235}U burnup is about 38% at BOEC and 50% at EOEC.

In changing from HEU fuel to LEU fuel, the Doppler coefficient increases, the coolant temperature coefficient decreases and the coolant density coefficient increases. The coolant void coefficient is also larger with LEU fuel than with the HEU fuels. These results are consistent with the harder neutron spectrum in the LEU core caused by a larger ^{235}U loading and additional ^{238}U. The sum of the temperature coefficients for fresh fuel loadings in each of the cores is about 11.5e-5 $\Delta k/k^2$/°C. It is also noted that the estimated reactivity coefficients with burned fuel at BOEC are larger than the coefficients with fresh fuel. The use of the smaller reactivity coefficients with fresh fuel will yield conservative results in the transient analyses.

Kinetic Parameters

The kinetic parameters for the BRR with the HEU and LEU fuels in fresh- and burned-fuel core configurations are shown in Table 4.

Table 4. BRR Kinetic Parameters (ENDF/B-VI data)
(All fresh fuel, 228-assembly core configurations with fresh-fuel cross sections.)

Parameter	HEU-UAl 40.2 g ^{235}U	$HEU-UO_2$ (BOEC)[a] 44.3 g ^{235}U	$LEU-UO_2$ 52.3 g ^{235}U
β-effective	7.333e-3	7.337e-3 (7.375e-3)	7.331e-3
Prompt neutron generation time, μs	49.1	46.5 (56.0)	41.5

[a] Equilibrium core configuration at BOEC with burned-fuel (32.5% ^{235}U burnup) cross sections; core-average ^{235}U burnup is about 38% at BOEC and 50% at EOEC.

The values of β-eff are nearly identical in the three cores with fresh fuel. β-effective is not sensitive to the neutron spectrum. The prompt neutron generation time becomes shorter as the ^{235}U content per assembly is increased because the neutron spectrum becomes progressively harder. Likewise, since the neutron spectrum becomes softer with ^{235}U burnup, the prompt neutron generation time is larger in a BOEC core than in a fresh core.

Control Rod Worths

In the analysis of BRR transients, the reactivity worths of the safety rods are necessary in order to estimate reactivity shutdown rates upon scram. These reactivity worths were calculated relative to the critical, banked-rod position of the control rods with the regulating and safety rods withdrawn. Table 5 indicates whether the safety rod is withdrawn (W) or inserted (I). The reactivity worth of safety rod pairs and the total worth of the three safety rods were calculated for BOEC core configurations with the HEU and LEU fuels.

Table 5. Critical Control-Rod Position
And Reactivity Worths of Safety-Rods at BOEC

Rod	HEU-UAl 40.2 g ^{235}U	$HEU-UO_2$ 44.3 g ^{235}U	$LEU-UO_2$ 52.3 g ^{235}U
CR Position[a], cm	25.6	24.8	22.5
SR Worth[b], % $\Delta k/k^2$			
SR-1(W) SR-2(I) SR-3(I)	4.69	4.79	4.78
SR-1(I) SR-2(W) SR-3(I)	4.49	4.56	4.47
SR-1(I) SR-2(I) SR-3(W)	4.64	4.71	4.61
SR-1(I) SR-2(I) SR-3(I)	7.23	7.41	7.31

[a] SR's and RR withdrawn to 5 cm above the core and each CR withdrawn to the critical rod position from 5 cm below the core. With the 60-cm core height, the fully withdrawn rod position is 70 cm and the fully inserted position is 0 cm.
[b] Worths are relative to SR's and RR withdrawn and CR's withdrawn to the critical rod position.

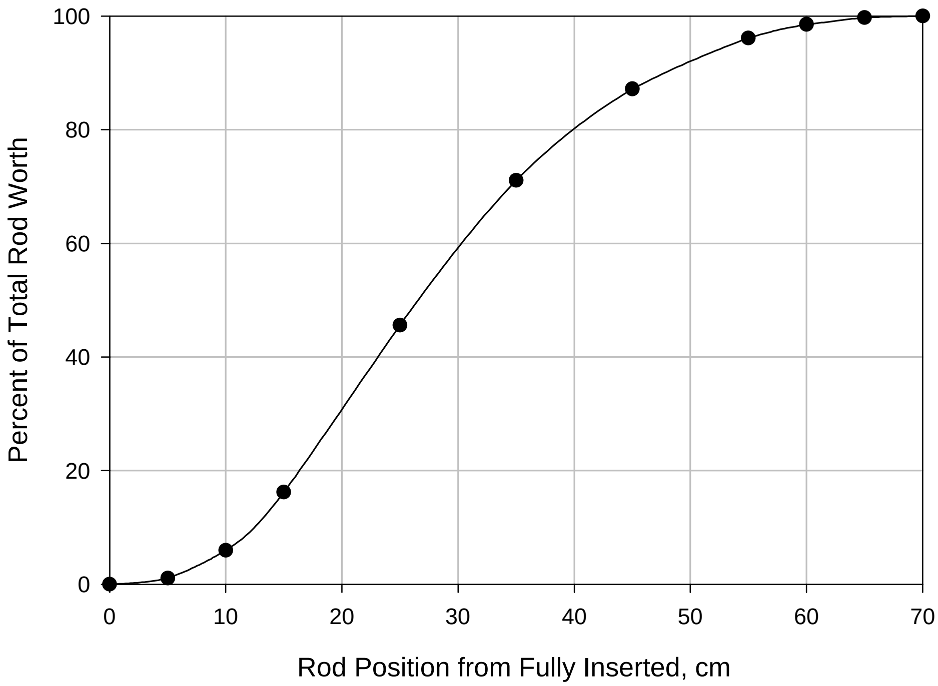

Figure 2. Safety-Rod Reactivity Worth Curve.

The critical, banked control-rod position is about 25 cm in the HEU cores and 22.5 cm in the LEU core. The safety rod worths are similar in the three BOEC core configurations. Slight differences in the rod worths are present due to core asymmetries. With the largest worth rod (SR-2) withdrawn, the combined worth of SR-1 and SR-3 is about 4.5% $\Delta k/k^2$. The calculated total worth of the three safety rods is about 7.3% $\Delta k/k^2$. Figure 2 shows the shape of the calculated worth curve of a safety rod over its full 70-cm stroke.

Representative Reactor Transients

The RELAP5 and PARET codes were used to perform transient analyses for cores with both HEU and LEU fuels. The RELAP5 model explicitly models the four coolant channels and three fuel elements (as parallel plates) in the VVR-M2 fuel assembly. One fuel assembly is the hot fuel assembly and the other fuel assemblies in the core are combined as an average fuel assembly. The PARET model uses one fuel plate and one coolant channel to represent the fuel assembly. One channel is the hot channel and the remaining channels in the core are combined to form an average channel.

Three transients for the BRR with HEU and LEU fuels were analyzed to compare reactor responses with the different fuels. The reactor transients were defined in Ref. 14 and include fast and slow reactivity insertion, and loss-of-flow transients from full power (10 MW) with coolant flow rates per assembly of 7.5 m^3/s. The fast and slow reactivity insertion transients assume trips at 120% of normal power and the loss-of-flow transient assumes a trip at 70% of normal flow. A 100 ms trip delay is assumed for all transients. The fast reactivity insertion rate was 1.15% $\Delta k/k^2$ in 0.8 s and the slow reactivity insertion rate was 2.7% $\Delta k/k^2$ in 5 s.

For these transients, the safety rod worths shown in Table 5 were used to scram the reactor. These safety rod worths with HEU fuel (36.2% enr., 44.3 $g^{235}U$) and LEU fuel (19.75% enr., 52.3 $g^{235}U$) are 4.56% $\Delta k/k^2$ and 4.47% $\Delta k/k^2$, respectively. The insertion time of the safety rods was 0.5 s.

Transient analyses were performed for a 228-assembly core with HEU and LEU dispersion-fuels, and kinetic parameters from Table 4 and reactivity coefficients from Table 3.

Figure 3 shows the RELAP5 plots of the peak fuel, peak clad and peak coolant temperature, and the peak reactor power for the fast reactivity insertion transient with HEU and LEU fuel. Similar data for the slow reactivity insertion and the loss-of-flow transients are shown in Figs. 4 and 5, respectively. These results show that the transient behavior is similar with either HEU or LEU fuel. The peak temperatures for the LEU fuel are only a few degrees higher than for the HEU fuel, and in both cases the temperatures are low and well within acceptable (safe) limits.

Analyses of the transients showed good agreement using both the RELAP5 and the PARET codes. Transient sensitivity analyses for the HEU fuel were also performed to estimate the effect of safety-rod worth (4.56% vs. 3.6% given in Ref. 14) and fuel meat thermal conductivity (178 W/m/°C vs. 87 W/m/°C given in Ref. 14). The results of these sensitivity analyses show only small differences in the peak temperature and power predictions.

CONCLUSIONS

An initial fuel transition sequence was calculated to convert the BRR from 36% enriched HEU fuel to 19.75% enriched LEU fuel. This sequence yielded excess reactivities that were too large in the last few cycles of the transition because of the larger lifetime of the LEU fuel assemblies and achievement of the ^{235}U burnup limit of the HEU (36%) fuel assemblies. A final conversion plan needs to be developed taking into account the final LEU fuel specifications and transition core loading sequence. With 6 - 8 transition cores, the reactor would be converted from HEU to LEU fuel in 3 - 4 years.

Equilibrium fuel cycle characteristics were compared for 19.75% enriched UO_2-Al dispersion fuel with 50.0 and 52.3 g^{235}U per assembly. With the LEU assembly containing 50.0 g^{235}U, three additional LEU fuel assemblies would be used per year compared to fuel assemblies containing 52.3 g^{235}U. Irradiation tests for the equivalent 50.0-g^{235}U fuel assemblies are currently in progress at the Petersburg Nuclear Physics Institute in Russia.

Safety parameters and transient analyses were calculated for the BRR with both (36%) HEU and (19.75%) LEU fuel. The differences in the safety parameters between the HEU and LEU cores are due to the higher ^{235}U content and harder neutron spectrum in the LEU core. Reactor transients with HEU and LEU fuel are very similar. Peak fuel, clad and coolant temperatures for fast and slow reactivity insertion transients and loss-of-flow transients are easily within an acceptable range. Control- and safety-rod-worths are sufficient to meet all reactor shutdown margin requirements. Overall, these studies indicate that the BRR can be safely and efficiently converted to LEU fuel when this fuel has been irradiation tested.

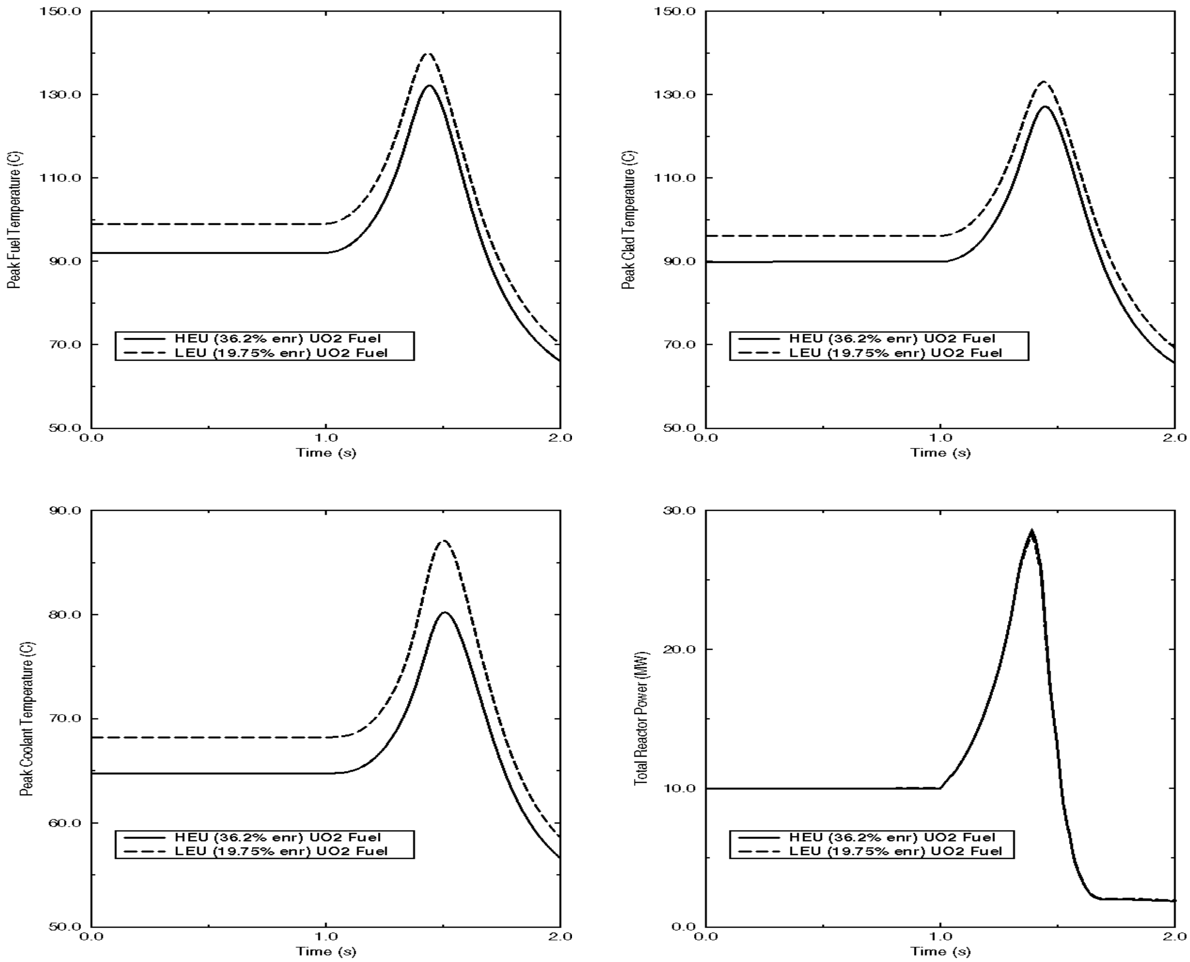

Figure 3. Fast Reactivity Insertion Transient (Fuel, Clad, Coolant, Power).

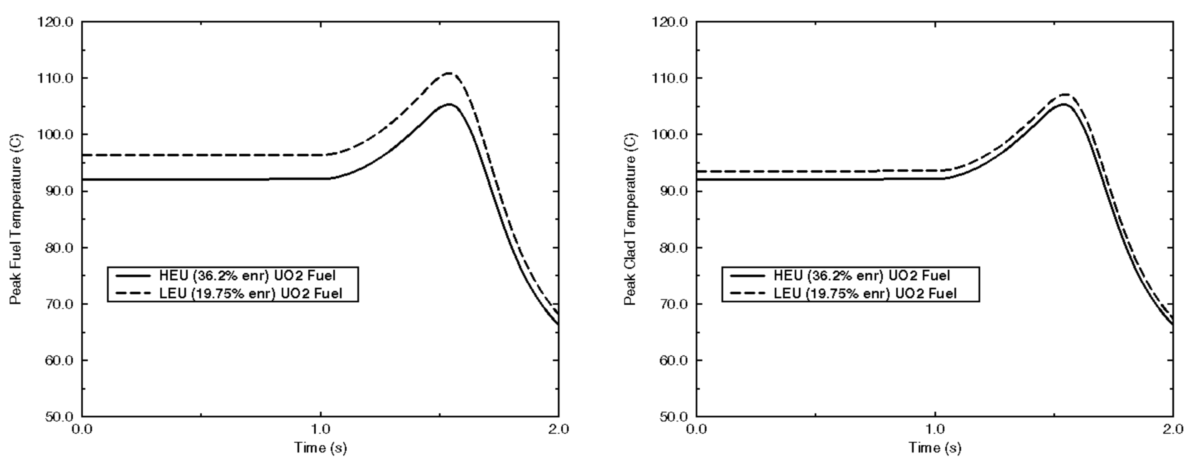

Figure 4. Slow Reactivity Insertion Transient (Fuel, Clad).

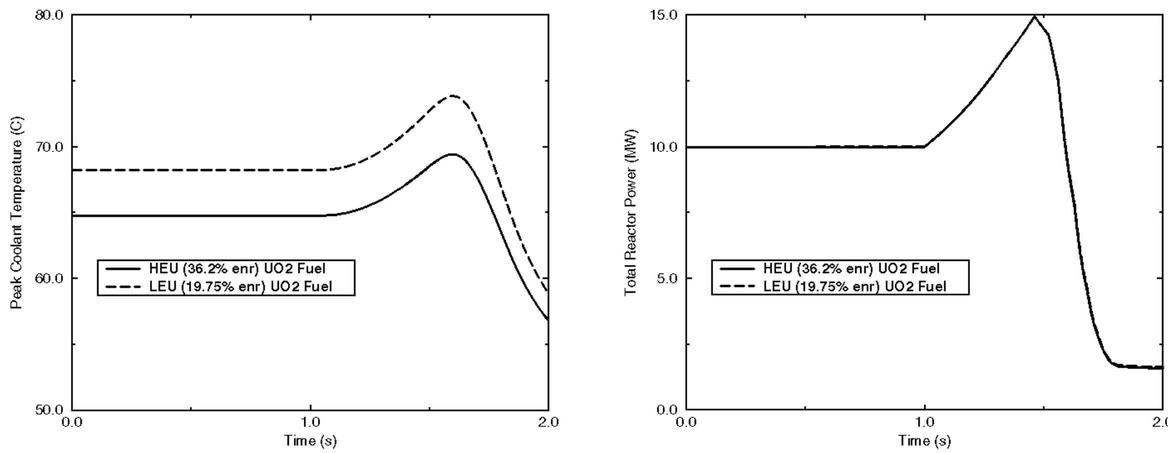

Figure 4. Slow Reactivity Insertion Transient (Coolant, Power).

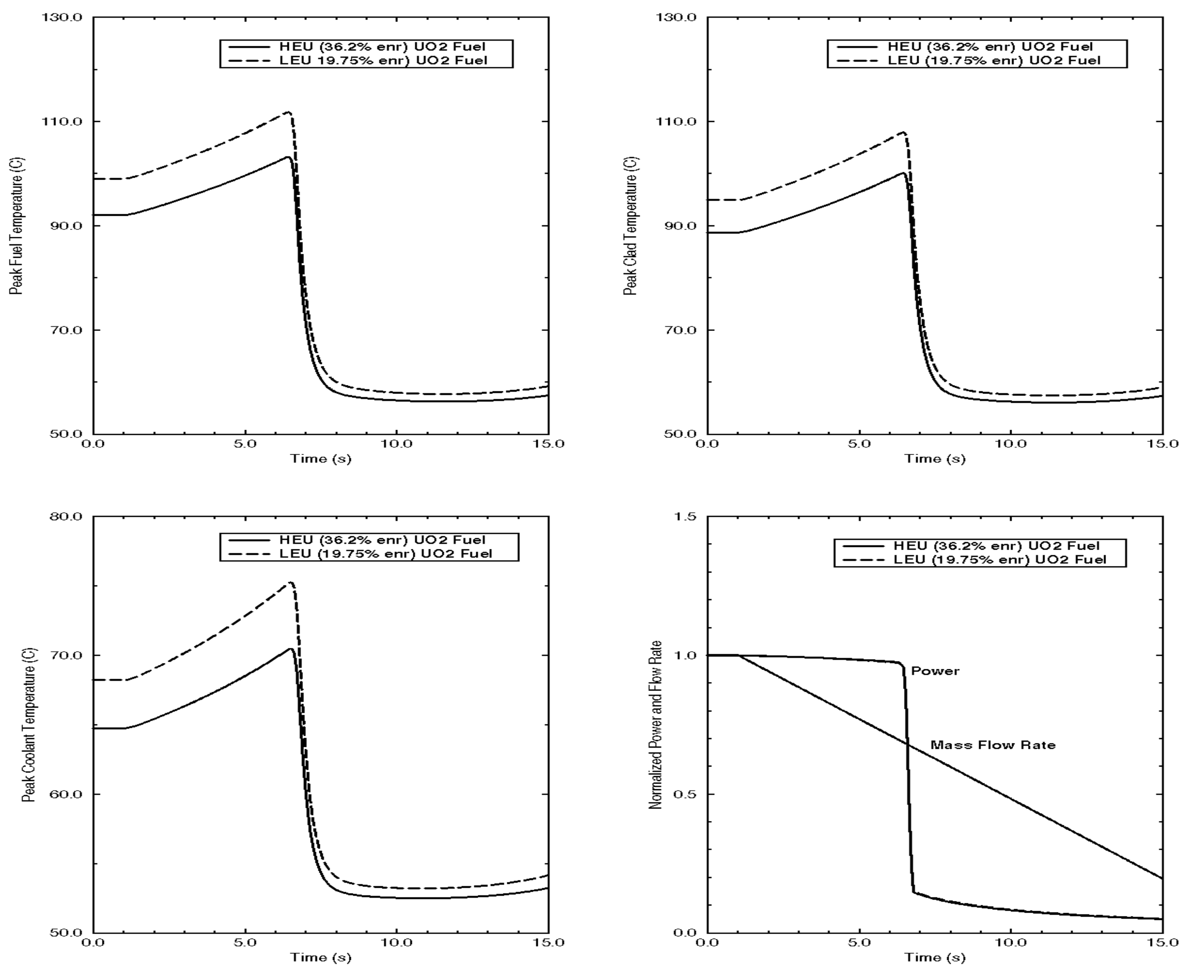

Figure 5. Loss-of-Flow Transient (Fuel, Clad, Coolant, Power).

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